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STRUCTURAL DESIGN CRITERIA FOR TARGET/BLANKET SYSTEM COMPONENT MATERIALS FOR THE ACCELERATOR PRODUCTION OF TRITIUM PROJECT

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ABSTRACT

The design of target/blanket system components for the Accelerator Production of Tritium (APT) plant is dependent on the development of materials properties data specified by the designer. These data are needed to verify that component designs are adequate. The adequacy of the data will be related to safety, performance, and economic considerations, and to other requirements that may be deemed necessary by customers and regulatory bodies. The data required may already be in existence, as in the open technical literature, or may need to be generated, as is often the case for the design of new systems operating under relatively unique conditions.

The designers' starting point for design data needs is generally some form of design criteria used in conjunction with a specified set of loading conditions and associated performance requirements. Most criteria are aimed at verifying the structural adequacy of the component, and often take the form of national or international standards such as the ASME Boiler and Pressure Vessel Code (ASME B&PV Code) or the French Nuclear Structural Requirements (RCC-MR). Whether or not there are specific design data needs associated with the use of these design criteria will largely depend on the uniqueness of the conditions of operation of the component. A component designed in accordance with the ASME B&PV Code, where no unusual environmental conditions exist, will utilize well-documented, statistically-evaluated developed in conjunction with the Code, and will not be likely to have any design data needs. On the other hand, a component to be designed to operate under unique APT conditions, is likely to have significant design data needs.

Such a component is also likely to require special design criteria for verification of its structural adequacy, specifically accounting for changes in materials properties which may occur during exposure in the service environment. In such a situation it is common for the design criteria and design data needs to evolve as the design progresses, operating conditions are refined, and materials characteristics in the unique environment are established.

This paper develops the relationship between the designers' data needs and the structural design criteria recently adopted for the Target Blanket System of the APT. The latter, the newly-developed APT Supplemental Structural Design Requirements (APT SSDR), was patterned after the design criteria developed for the International Thermonuclear Experimental (Fusion) Reactor (ITER). A summary description of the design rules based on the APT SSDR is presented, and the impact of these rules of changes in materials properties resulting from exposure in the APT proton/neutron irradiation environment are discussed.

I. INTRODUCTION

A. Description of APT Plant and Target/Blanket Assembly

The Accelerator Production of Tritium (APT) project is part of a Department of Energy (DoE) dual track strategy to supply tritium to support the long-term needs of the nation's nuclear weapons stockpile. Tritium decays at a rate of 5.5% per year, and must be replaced on a regular basis. The APT Target/Blanket (T/B) system is comprised of a T/B assembly and attendant heat removal systems. The T/B assembly produces tritium using a high-energy proton beam and a spallation neutron source. All systems reside within a T/B building located at the end of a linear accelerator. During operation, protons are accelerated to an energy of 1030 MeV at a current of 100 mA. The protons interact with tungsten and lead nuclei in the T/B assembly to produce neutrons through the process of nuclear spallation. Neutron capture in ^{3}He gas contained within tubes in blanket components surrounding the tungsten neutron source (target) produces tritium which is removed on a semi-continuous basis. All T/B components operate at low temperature in a mixed proton/neutron environment, and are cooled with water. The T/B assembly is modular to allow removal and replacement of components. Structures directly in the proton beam, which endure the highest irradiation damage, are replaced annually. Materials in the blanket region are primarily in a low-neutron flux environment and potentially will last the life of the facility.

The APT T/B system is housed in a building located at the end of a 1.3-km-long linear accelerator. The proton beam enters the building through a high-energy beam transport (HEBT) system operating at a vacuum level of 1 x 10^{-5} torr which expands the beam. The expanded beam is directed onto the T/B assembly which is housed within a large vessel operating at a vacuum level of 1 torr. The pressure boundary between the vessel and HEBT is a proton beam window through which the expanded beam passes. Figure 1 shows an isometric view of the T/B assembly showing the major components. These include the proton beam window, the tungsten neutron source, the neutron decoupler, the blanket and reflector components, the upper vessel internals, and cooling water and gas-handling systems. The T/B system is housed within a stainless steel vessel 9.75 m in diameter. Steel shielding between the T/B assembly and vessel wall keeps radiation damage to the vessel at a minimum.

The proton beam window is a double-walled, water-cooled structure separating the high-vacuum environment of the beam expander from the rough-vacuum environment of the T/B cavity vessel. The window operates at a peak temperature of 155°C . The tungsten neutron source (TNS) consists of nested, heavy water-cooled metal clad tungsten cylinders and rods assembled in horizontal metal tubes. The horizontal tubes are manifolded into larger diameter vertical inlet and outlet pipes like the rungs of a ladder. The peak operating temperatures of the components in the TNS are 198°C (tungsten) and 164°C (clad surface temperatures).

High-energy particles scattered out of the TNS leak into surrounding blanket modules, after passing through a decoupler region immediately surrounding the TNS. The decoupler region consists of several rows of tightly-packed ^{3}He-containing tubes contained within a multi-channel extruded structure, with light water coolant flowing in the annuli between the tubes and channels in the extrusion. Peak operating temperatures in the decoupler tubes is 62°C.

Blanket components surround the TNS and decoupler, and consist of rows of tightly-packed ^{3}He-containing tubes in a multi-channel extruded structure similar to that of the decoupler. Unlike the decoupler, the structure contains channels for containment of lead to provide an additional source of neutrons from additional spallation from the lead as well as channels for water cooling of the ^{3}He tubes. The peak operating temperature of the blanket tube components is 107°C.

The blanket is surrounded by a reflector region containing water-cooled ^{3}He tubes in an extruded structure similar to that of the decoupler. Steel shielding surrounds the blanket and reflector to minimize activation of the vessel and external structures and to protect workers. In addition, shields above the T/B region minimize activation of the upper vessel structures. The first 100 cm of shielding surrounding the blanket and reflectore requires active cooling by means of light-water cooling panels mechanically attached to the shield blocks.

The upper vessel houses a number of structures providing all of the utilities required to operate the T/B modules. This includes headers for heavy-water and light-water coolant, connecting piping from the headers to the modules, ^{3}He gas line connections, and instrumentation lines.

Encasing the T/B assembly and shielding is a sealed stainless steel pressure vessel. The vessel is cylindrical in shape with a removable head structure for access and extraction of internal components. It provides a vaccum atmosphere for the beam to pass through and is also a confinement barrier and radionuclide barrier in the event of an internal coolant leak or ^{3}He gas line leak. To provide a backup heat removal mechanism during postulated accidents, the cavity vessel can be flooded from an adjacent storage pool.

B. Description of T/B Assembly Structural Materials and Operating Environment

Table 1 shows a summary of the candidate structural materials for the T/B assembly, and the nominal operating environment for each T/B assembly component. In addition to the conditions shown in the table, a potentially-corrosive water coolant environment exists for the materials, especially for materials directly in the proton beam, resulting from coolant radiolysis and sensitization of the material surfaces by the charged particle environment (Ref. 1).

II. DESIGN OF APT TARGET/BLANKET SYSTEM COMPONENTS

A. Design Data Needs

The design of components for any complex system is often dependent on the development of materials properties data specified by the designer, i.e., the designers' design data needs (designated DDNs by the APT project). Such design data are generally directly related to the design criteria being used and a specified set of operating conditions (e.g., loads, temperatures, etc.) and performance requirements. Most design criteria are aimed at verifying the structural adequacy of components and may take the form of national or international standards or codes, such as the American Society of

Mechanical Engineers Boiler and Pressure Vessel Code (ASME B&PV Code) or the French Nuclear Structural Requirements (RCC-MR). Requirements for DDNs are generally associated with the use of specific design criteria and largely depend on the uniqueness of the conditions of operations of the operating equipment. For example, components designed to Section VIII, Division 1 of the ASME B&PV Code (unfired pressure vessels) will use materials whose properties are specifically covered in the Code (Table 1, Section II, Part 2). However, components designed to operate under unique conditions like that imposed for the APT T/B assembly components (e.g., proton/neutron irradiation, temperature, corrosive media, etc.) are likely to have significant DDNs.

The general methodology for developing design data needs for any system includes the following items:
1) Development of a general knowledge of the operating system environment and operations parameters, terms, issues, etc.; 2) Definition of expected required materials properties data/issues based on expected environmental effects (e.g., irradiation, corrosion, etc.) and the review of adequacy with the designers;
3) Development of operational parameters/service conditions for each component (normal and off-normal operation) including chemical environments (temperature, pressure, irradiation fluence, dpa, He, H, etc.).; and 4) Preparation of design data needs which reflect all of the above.

The general methodology for developing design/materials data relative to the developed design data needs includes the following: 1) Selection of a specific design criteria, which includes required materials properties [Relatively easy if dealing with well-characterized structures in well-characterized environments (e.g., a pressure vessel operating in air, for example)]; 2) Definition of the operation history of the component [temperature, pressure, other loads (monotonic, cyclic), other environmental influences (Flu-ence, dpa, etc.)]; 3) Selection of applicable candidate structural materials; 4) Development of test specifications and plans for obtaining those data that are not currently available (from open literature, handbooks, etc.); 5) Acquisition of materials properties data (via literature and/or experiment/test); and 6) Utilization of data with the rules defined in the structural design criteria for component design and verification of structural adequacy.

A typical set of materials properties data which will be required for use with the structural design criteria will include: 1) Tensile Properties (yield Strength, ultimate strength, uniform elongation, reduction of area, elastic modulus, Poisson's ratio, and stress/strain curves); 2) Fracture Toughness; 3) Low- and High-Cycle Fatigue Curves and Fatigue Crack Growth; 4) Irradiation Creep; 5) Thermal Creep and Thermal Creep Crack Growth; 6) Stress Relaxation; 7) Thermal Properties (thermal conductivity, thermal expansivity, emissivity, specific heat); and 8)Physical Properties (melting temperature, density or irradiation-induced wwelling, change in material composition)

A. Degradation of Materials in the APT Environment

The operational and environmental conditions associated with the APT T/B system present a substantial number of materials-related issues that must considered in its design. These include the effects of proton and neutron irradiation on material properties, the effects of corrosion (heavy water, light water, and air) on component material wastage, the production and behavior of various key transmutation species (He, H, and Hg), the production, implantation, retention, and permeation of tritium in ^{3}He blanket tubes, and key interface reactions (e.g., adhesion of tungsten target cladding materials). These issues have prompted the development of substantial APT T/B system DDNs to ensure that the designer has adequate information to apply to the structural design criteria.
Perhaps the major issue confronting APT materials specialists and designers is the effect of proton and neutron irradiation on materials design properties. Neutrons are produced in the T/B assembly by nuclear spallation of the tungsten and lead. These neutrons, together with the protons, produce irradiation damage in the T/B assembly components. The degree of damage produced depends on the neutron and proton fluences imposed on the component (by virtue of the relative position of the component with respect to the proton beam and target) and on the materials of construction used. Proton and neutron irradiation produces significant material damage by displacement of atoms from their normal atom sites and via the production of hydrogen and helium within the material. This irradiation damage is known to produce either increases or decreases in strength, and can produce severe losses in ductility and fracture toughness. Figures 2, 3, and 4 show proton/neutron irradiation data obtained for 316L stainless steel (316L SS) from an irradiation test performed under near-APT irradiation conditions in the Los Alamos Nuetron Science Center (LANSCE) irradiation test facility (Refs. 2,3).

Figure 2 shows the substantial increase in tensile yield strength of 316L SS with proton/neutron irradiation (calculated displacements per atom, dpa), while Figure 3 shows the corresponding decrease in ductility experienced by the material. At ~3 dpa, for irradiation in the temperature range indicated, 316L SS achieves a materials damage state which results in a plastic instability in the material very early in its stress/strain

behavior, producing very low elongation values. A corresponding decrease in
the fracture toughness of 316L SS with irradiation is shown in Figure 4.

B. Materials degradation beyond applicability of standard design codes

The degradation of materials properties, especially ductility, by high-energy particle irradiation, provides a substantive issue for the use of standard design codes in the design of components operating in such environments. For components which are designed for use under conditions where no unique environmental conditions exist, standard design codes, such as the ASME B&PV Code, can be utilized for conservative component design. The ASME Code design rules have been developed assuming that materials exhibit elastic-perfectly plastic behavior. Although not specifically stated in the Code, the assumption is based on the requirement that acceptable materials must be capable of significant plastic flow before failure. A non-stated, "rule of thumb" which has been utilized for many years, suggests that relevant materials should possess 5 to 10% ductility for applicability with respect to design by the Code.

On the other hand, components designed for use under unique environmental conditions, such as those experienced by the APT T/B system, where materials may suffer extreme losses in ductility, will require the use of special design criteria for verification of their structural adequacy.

III. FUNDAMENTAL BASIS FOR APT STRUCTURAL DESIGN CRITERIA

To face the constraints on the applicability of standard design codes for materials operating in the unique APT environment, the APT project has adopted a structural design criteria which combines the use of the ASME Code with supplemental structural design requirements derived from the design criteria developed for the International Thermonuclear Experimental Reactor (ITER) (Ref. 4). The supplemental ITER Structural Design Criteria (SDC) were developed by an international group of materials and mechanical design experts from the USA, France, Japan, Germany, and Russia, and include elements primarily from the French fast breeder reactor codes (RCC-MR), incorporating in them, whenever applicable, rules from the other ITER partners (USA – ASME Code Case N-47), Japan (Miti Notification N501, Monju Guide) and Russia (PNEAG-7-00-2-89). As with all design rules, the primary objective of the ITER SDDC was to ensure that required safety margins were maintained relative to the types of mechanical damage which might occur as a result of imposed loading during operation. The second key step was to develop specific rules for irradiation effects on materials properties. These include irradiation-induced creep and swelling, and irradiation-induced changes in materials properties [hardening, embrittlement, nuclear transmutations and gas formation (He and H).

The basic concept for the APT Structural Design Criteria utilizes the ASME B&PV Code, Section VIII, Division 2, for vessel components, and the ANSI/ASME Piping Code, B31.3 Petroleum, for piping, hereafter referred to as the "Code". The Code is supplemented with APT Supplemental Structural Design Requirements (SSDR) (Ref. 5) which are based on those elements of the ITER SDC which specifically treat irradiation-induced changes in materials properties beyond the applicability of the Code. The APT SSDR rules provide for a smooth transition from Code rules to supplemental rules , and ensure a margin of safety equal or greater than that in the Code for similar loading phenomena. The rules provide for a "design by analysis" approach, which is consistent with Section VIII, Division 2, of the Code. The design by analysis may use elastic analysis, which generally provides for limiting stresses, or inelastic analysis, which provides for limiting strains. The type of analysis, elastic or inelastic, is left to the discretion of the designer except for some cases where certain strain limits are exceeded and require inelastic analysis.

The APT SSDR addresses a number of failure modes including monotonic failure modes (plastic collapse, plastic flow localization, local ductility exhaustion, and fast fracture), cyclic failure modes [progressive deformation, and fatigue (including crack initiation and propagation)], and buckling (load controlled or displacement controlled). The SSDR also addresses both low temperature and high temperature (creep regime) rules. In conjuction with these failure modes, the SSDR specifically addresses irradiation effects on materials properties including low ductility, fracture toughness, plastic instability (lack of work hardening), plastic flow localization, ductility exhaustion, irradiation-induced creep, and irradiation-induced swelling.

The nomenclature used for stress, strain, loading, etc., in the APT SSDR is the same as that used in the Code. Three categories of stress are defined: Primary stresses include imposed stresses which are not affected (i.e., relaxed or decreased) by small displacements, such as the deadweight of components or pressure. Secondary stresses are that portion of the total imposed stress, excluding peak stresses, which can be relaxed by small displacements and are self limiting, such as thermal constraint and swelling constraint. Peak stresses are that portion of the total imposed stress produced by non-linear stress distribution and stress concentrations, which are localized and cannot generally cause overall structural deformation.

Loads applied to the structure are grouped into three levels (catagories) depending on the amount of structural damage they are allowed to inflict: Load Level A includes loads from operation and likely events and produces negligible component structural material damage; no component inspection is required for continued operation. Load Level C includes loads from unlikely events and may produce significant distortion in a component material; inspection may be required for continued operation. Load Level D includes loads from extremely unlikely events and may produce significant general distortion in a component, but with no loss of safety function. Elastic and/or inelastic safety factors specific to each load level are applied with respect to maximum stress or strain limits for each type of failure mechanism to provide safety margins consistent with or greater than those specified by the Code.

V. APT SUPPLEMENTAL STRUCTURAL DESIGN REQUIREMENTS (RULES)

The APT SSDC provides rules for specifying limits based on both stress and strain depending on whether the designers desire to apply analyses based on stress or strain. In the following, only limits based on stress are discussed. Similar limits based on strain are also included in the APT SSDR.

A. Primary Membrane Stress Limits (Primary Plastic Collapse), S_m

S_m is a basic design allowable stress used to guard against monotonic types of damage having consequences such as immediate excessive deformation (beyond purely elastic behavior) and immediate plastic instability (e.g., necking during tension). S_m is used for basic thickness calculations to limit primary membrane stresses (as in the ASME Code) and to prevent primary plastic collapse of a structure. The values of S_m are obtained by applying specific coefficients or reduction factors to values of the minimum tensile yield strength (Sy_{min}) and minimum ultimate tensile strength (Su_{min}) for unirradiated and irradiated material at room temperature and elevated temperatures, and taking the minimum value obtained as S_m. These coefficients range from 1/3 of Su_{min} and 2/3 of Sy_{min} for materials other than annealed austenitic stainless steels, and from 1/3 – 1/2.7 Su_{min} and 2/3 – 0.9 Sy_{min} for austenitic stainless steels, with values depending on temperature, irradiation level, and strain hardening capability of the material. For materials which strengthen on irradiation, S_m is a specified fraction of the minimum unirradiated tensile yield or ultimate strength. For materials which soften or lose strength on irradiation, S_m is a specified fraction of the minimum irradiated yield or ultimate strength for the maximum irradiation dose achieved during the intended life of the component material.

B. Primary Local Membrane and Primary Local Membrane plus Bending Stress Limits (Local Plastic Collapse), K_{eff}

Because irradiation may significantly reduce the work hardening capability and uniform elongation of a material, a reduction in the normal ASME B&PV Code K-factor (of 1.5) for defining the limit for primary local membrane stresses and primary local membrane plus bending stesses to prevent local plastic collapse is introduced depending on the modified shape of the stress/strain curve. This factor is a function of minimum irradiated tensile yield and ultimate strengths, minimum irradiated uniform elongation, and the shape of the cross-section, as follows (for a rectangular cross-section) :

$$K_{eff} = K_{rect}\{[3-\gamma^2 + (\beta - 1)(1 - \gamma)(2 + \gamma)]/(1 + 2\beta)\} \quad (1)$$

where Ke_{ff} = 1.5, $\gamma = [Sy_{min}(irr)/(Su_{min}(irr) + E\varepsilon_{umin}(irr))]$, $\beta = Su_{min}(irr)/Sy_{min}(irr)$, $Sy_{min}(irr)$ and $Su_{min}(irr)$ = minimum irradiated tensile yield and ultimate strengths, E = Young's Modulus, and $\varepsilon_{umin}(irr)$ = minimum irradiated uniform elongation. For irradiated materials with ε_u values decreasing to low levels (1%), K_{eff} decreases on irradiation, falling from a value of 1.5 to values approaching ~1.35.

C. Primary and Secondary Membrane Stress Limits (Plastic Flow Localization), S_e

In the ASME B&PV Code, combined primary and secondary stresses are allowed to reach values where some yielding of the material may take place. This is derived from the fact that, for materials with significant ductility, yielding of the material will generally relieve secondary loadings. However, when subjected to irradiation, ductility may be significantly reduced, and materials may reach a point where the material has insufficient ductility to redistribute strain in a structure and decrease stresses. In this situation even secondary membrane stresses can cause failure. This is particularly important when an irradiated material loses its strain hardening capability. Under this circumstance, its uniform elongation reaches very low values (less than a few percent), and consequently, its yield and ultimate strengths become nearly equal. To guard against such situations, and to provide adequate margin against the rapid spread of plasticity through the thickness of an irradiated material because of its reduced work hardening capability, an additional limit, S_e, is placed on total membrane stresses arising from both primary and secondary stresses. Using a method proposed by the French (Ref. 4), the ITER SDC (and APT SSDR) provide a limit on combined primary and secondary membrane stresses that is a function of minimum irradiated ultimate tensile strength and minimum irradiated uniform elongation, as follows:

$$S_e = \beta_1[Su_{min}(irr) + E(\varepsilon_{umin}(irr) - 0.02)/2r_1] \qquad (2)$$

where β_1 is a factor which varies depending on the load level (0.33 for level A, 0.4 for level C, or 0.67 for level D), $S_{umin}(irr)$ is the minimum irradiated ultimate tensile strength, E is Young's Modulus, $\varepsilon_{umin}(irr)$ is the irradiated minimum uniform elongation, and r_1 is the elastic follow-up factor (maximum of 4 for $\varepsilon_{umin}(irr)>0.02$ and infinity for $\varepsilon_{umin}(irr)<0.02$). The value of 2% minimum uniform elongation ($\varepsilon_{umin}(irr) = 0.02$) has been determined (by ITER design experts) to represent that value of strain which is expected to sufficiently redistribute (and reduce) stresses in a loaded structure.

D. Total Local Stress Limits (Local Fracture), S_{d1} and S_{d2}

Safety margins must also be introduced to prevent localized failure (i.e., crack initiation) due to loss of ductility of irradiated materials. To provide an adequate margin against crack initiation under monotonic loading at a local high stress point (e.g., notch), additional limits, S_{d1} and S_{d2}, are applied to total local stresses, including local primary and secondary membrane stresses + bending stresses + peak stresses (S_{d1}) and local primary and secondary membrane stresses + bending stresses (i.e., without peak stresses, S_{d2}). These limits are functions of minimum irradiated ultimate tensile strength and true strain to failure, which is a function of reduction of area measured in a tensile test, as follows:

$$S_d = \beta_2[Su_{min}(irr) + (E\varepsilon_{trmin}(irr)/r_1TF)] \qquad (3)$$

where β_2 is a factor which varies depending on the load level (0.67 for level A, 0.8 for level C, or 0.9 for level D), $Su_{min}(irr)$ is the minimum irradiated ultimate tensile strength, E is Young's Modulus, $\varepsilon_{trmin}(irr)$ is the minimum irradiated true strain to failure = $\ln\{100/[100 - RA_{min}(irr)]\}$ and $RA_{min}(irr)$ = % minimum irradiated reduction in area, r_1 is the elastic follow-up factor [maximum of 4 for $\varepsilon_{umin}(irr)>0.02$ and infinity for $\varepsilon_{umin}(irr)<0.02$ when peak stress is not included in S_d (i.e., Sd_2), and equal to the elastic stress concentration factor , K_T, or a maximum of 4, when peak stress due to a stress concentration is included in S_d (S_{d1})], and TF is a triaxiality factor defined by the shape of the stress point (maximum of ~3 for a sharp notch). The key parameter for this stress allowable is the true strain to failure, that local ductility value above which the material will initiate a crack. It should be noted that the S_d requirement is in addition to that required to demonstrate acceptability with respect to cyclic fatigue.

E. Fast Fracture Limits (Critical Flaw Size), K_I

Generally, ductile materials also possess high fracture toughness, and as long as a material's toughness remains sufficiently high (or the relevant stresses sufficiently low) no additional rule is needed to guard against sudden fracture. However, fracture toughness values generally decrease substantially with increasing irradiation dose. As a general rule, in order to avoid sudden fracture the stress intensity factor, K_I, imposed in a material under the operating loads must remain less than a critical stress intensity, K_{Ic}, (the fracture toughness) for the material, that is, one has to justify the ability of the component in the presence of a postulated flaw to withstand, with an adequate safety margin, the expected loads without undergoing brittle fracture. The size of the postulated defect is generally determined by the resolution capability of the non-destructive examination method utilized to determine the size of defects present in as-fabricated components. The acceptability of this defect size must be periodically evaluated during the lifetime of the component using cyclic crack growth data. The stress intensity factor, K_I, used in the case of elastic analysis, can be approximately expressed in terms of the applied stress, σ, and the flaw depth, a, by

$$K_I = \sigma(\pi a)^{1/2} \qquad (4)$$

If the material retains some ductility, one can assume that the secondary stresses will be relaxed, and hence, the stress is limited by the ultimate tensile strength of the material. However, if the material ductility reaches very low levels such that secondary stresses cannot be assumed to be relaxed out, more stringent requirements must be imposed to guard against fast fracture emanating from an unobserved (by inspection) intitial defect (or propagated initial defect). The APT SSDR requires a calculation to ensure that there is sufficient margin between the value of K_I and K_{Ic}, the fracture toughness of the irradiated material. Cyclic crack growth in irradiated materials is also included in the determination of the defect size. The SSDR places specific limits on K_I versus K_{Ic} for primary and secondary membrane stresses and for primary + secondary membrane stresses + bending stresses + peak stresses. Utilizing this criterion, an allowable stress, S_I, to guard against sudden fracture by loss of toughness can be calculated by

$$S_I = K_{Ic}/(\pi k2a)^{1/2} \qquad (5)$$

where k2a = maximum of $4a_u$ or h/4, a_u = maximum undetected flaw size, and h = wall thickness of the component.

F. Fatique Stress Limits (Cyclic Crack Initiation)

Crack initiation due to fatigue is precluded by use of rules similar to those in the Code, but using fatigue curves modified to account for irradiation. In the ASME Code, a safety factor of 2 on strain and 20 on the number of cycles to failure are applied to fatigue data for unirradiated material. However, irradiation can have a significant effect on cyclic fatigue. In the low-cycle fatigue region (less than ~10,000 cycles), a material's ductility plays an important role, and with reduced ductility the fatigue life for a given strain range (or stress) is generally decreased. In the high-cycle regime, material hardening by irradiation (i.e., increase in ultimate tensile strength) generally increases the fatigue life while material softening generally lowers fatigue life. Figure 5 shows a schematic representation of fatigue behavior for a material that strengthens on irradiation and exhibits a loss of ductility. The APT SSDC utilizes the same factors of safety on strain and cycles to failure as does the ASME Code, but applies these to fatigue data for irradiated material as shown in Figure 5. For the case of a material which strengthens on irradiation, the fatigue curve is generally increased by irradiation. However, the APT SSCR does not allow the use of the higher fatigue values to establish allowable fatigue limits. In lieu of measured data on irradiated material, curves for unirradiated material may be modified according to the effects of irradiation on true strain to failure (function of reduction of area) and ultimate strength using a Manson/Coffin-type relation. Once the allowable fatigue stress has been determined, it must be compared to the value of S_d, the total local stress limit for local fracture due to ductility exhaustion, to determine the lower limit on total stress.

G. Limits on Progressive Deformation Under Cyclic Load Conditions (Ratcheting)

When a material is subjected to cyclic loading beyond its elastic limit, the overall permanent deformation may stabilize after a few cycles or may continue to increase with every cycle, resulting in progressive deformation, or ratcheting. Stress limits for progressive deformation are determined by using either the ASME Code "$3S_m$ rule", which ensures that the material shakes down to elastic action after the first few cycles, or the Bree Diagram rules set up in ASME Code Case N47-29. In the $3S_m$ rule, the maximum local primary membrane and bending stresses plus cyclic secondary and peak stresses are limited to $3S_m$. In the Bree Diagram rule, the cyclic secondary and peak stresses are limited to values based on the current level of primary stress. ITER has also considered the use of an "efficiency diagram rule" which defines a stress limit based on the definition of an effective primary stress that yields the same immediate deformation as that of the combined primary plus secondary stresses (Ref. 4)

H. Limits for Irradiation-Induced Swelling and Irradiation-Induced and Thermal Creep

The APT SSDC also provides guidance on how to account for irradiation-induced swelling, and irradiation-induced creep and thermal creep. Limits are provided for guidance as to when irradiation-induced swelling and irradiation-induced creep strains are considered negligible. Swelling is considered negligible when the change in volume per unit volume of material remains within 0.05%. If swelling is greater than this value, elastic analysis is required for assessing its impact on component stresses. Irradiation creep is generally considered to be a non-damaging phenomenon, and in addition, contributing to stress relaxation. However, as with other inelastic strains, irradiation creep must be considered for the evaluation of excessive deformation, fatigue, and ratcheting. Stress relaxation in bolts is an example where small deformations due to irradiation creep cannot be ignored. For irradiation-induced creep, the SSDC defines a negligible creep strain as that which is less that 0.05% linear strain for an applied stress (primary membrane + bending stress intensity) of 1.5Sm. A similar linear stain limit is applied for thermal creep, and thermal creep above this limit is treated in a similar way to that in ASME Code Case N-47. If the combined strains resulting from swelling and/or creep exceed 2%, inelastic analysis is required.

VI. IMPACT OF APT SSDR ON STRESS ALLOW-ABLES

The impact of the APT SSDR (rules) on the major design stress allowables for monotonic loadings are shown for two materials, 316L stainless steel and Alloy 718 (precipitation hardened), in Figures 6 and 7.

The material 316L SS is representative of a material which is strengthened by irradiation. As shown in Figure 6 for 316L stainless steel irradiated in the LANSCE facility at ~164C under near-prototypic proton and neutron irradiation conditions, the minimum tensile yield strength and ultimate strength of 316L SS both increase with irradiation (dpa), and approach each other at an irradiation level where the minimum uniform elongation value decreases to near zero (~3 dpa). It should be noted that although the uniform elongation falls to near zero values, values for the minimum true strain to failure, based on reduction of area values, remain substantial. The consequence of the strength increases and uniform elongation losses is to reduce the stress allowable for primary and secondary membrane stresses, S_e, from fairly large values (above 1000 Mpa) to values that are only approximately twice that for primary membrane stresses alone (S_m) for irradiation damage values above ~3 dpa. This places a real emphasis on minimizing secondary stresses, such as thermal stress. One should note that although irradiation increases the values of yield and ultimate strength, which would result in an increase in the calculated value of S_m with increasing irradiation dose based on one third of ultimate strength, the APT SSDC requires that the minimum calculated value of S_m be used for design, i.e., no credit can be taken for increases in strength in the calculation of the basic primary membrane stress allowable. Because the true strain to failure remains substantially high during irradiation, values of S_{d1} for limiting local primary plus local secondary membrane stress plus bending stress plus peak stress are very high (off the graph in Figure 6). However, the limit for local primary plus secondary membrane stress plus bending stress (i.e., without added peak stresses), S_{d2}, decreases from very high values to a limiting value at the point where the uniform elongation dereases to below a strain limit of 2%, reaching a value which is approximately twice that for the limit for combined primary plus secondary membrane stresses, S_e. Above ~3 dpa, calculated vales for S_e and S_{d2} increase slightly with irradiation, and are limited to one third and two thirds, respectively, of the minimum irradiated ultimate strength. These lower limits result primarily from the fact that the uniform elongation has decreased to it lowest level (neary zero), while the ultimate tensile strength is still increasing slightly.

Precipitation hardened Alloy 718 is an example of a material whose strength values (yield and ultimate tensile strengths) are reduced by irradiation. As shown in Figure 7, for Alloy 718 irradiated in the LANSCE facility at ~164C, the minimum yield strength for Alloy 718 first increases slightly with irradiation and then decreases while the minimum ultimate tensile strength for Alloy 718 remains fairly flat up to ~1 dpa and then decreases thereafter. For this material, irradiation reduces the uniform elongation to near zero at ~1 dpa, at which point the yield strength and ultimate tensile strength converge. Like 316L stainless steel, however, the reduction of area and true strain to failure remain at substantial levels. The consequence of these changes are that the stress allowable for primary and secondary membrane stresses, S_e, are reduced to that for primary membrane stresses alone, S_m, which is continually reduced by irradiation because of the irradiation-induced strength losses. There is, therefore, an increased emphasis over that for 316L stainless steel in minimizing thermal stresses, typically the largest of secondary-type stresses, for Alloy 718. Changes in the other stress allowables for total stress, S_{d1} and S_{d2}, are reduced in a manner similar to that observed for 316L stainless steel albiet these stress allowables decrease with increasing irradiation above the dpa value at which the uniform elongation approaches zero.

SUMMARY

The design of components for the target/blanket system of the Accelerator Production of Tritium (APT) facility will require the use of design stress allowables based on structural design criteria which specifically take into account the unique environment in which the structural materials for the APT plant will operate. The APT proton/neutron environment will produce substantial changes in mechanical properties which must be accounted for in component design. The incorporation of the effects of the APT environment on structural materials behavior has resulted in the development of specific data needs for T/B system component design. Structural design criteria have now been selected for APT T/B system component design that utilize the ASME Code, supplemented by structural design criteria patterned after those developed for the International Thermonuclear (Fusion) Experimental Reactor (ITER).

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Table 1
Candidate T/B Assembly Structural Materials and Their Nominal Operating Environment

Component	Primary Structural Material	Peak Structure Temperature (^{o}C)	Peak Irradiation Damage (dpa/y)[2]	Hydrogen Production (appm/y)[1]	Helium Production (appm/y)[1]
Beam Window	Alloy 718[3]	155	15	9400	750
Target/ Cladding	Tungsten/ Alloy 718[3,4]	198/164	36	27000	1600
Target Structure	Alloy 718[3,4]				
Decoupler/ Blanket/ Reflector Structure/ ^{3}He Tubes	6061 Al[5,4]	107	4.7	2600	150
Internal Shields	Clad Ferritic Steel[6,7]	75	0.63	850	24
Vessel Internals	316L Stainless Steel	60	0.1	21	3.6
Cavity Vessel	304L Stainless Steel	50	0.1	21	3.6

[1] "appm"= concentration, "atomic parts per million"
[2] "dpa" = calculated irradiation damage parameter, atom "displacements per atom"
[3] annealed or precipitation-hardened condition
[4] alternate candidate material is 316L stainless steel
[5] aged-hardened (-T6) condition
[6] normalized and tempered condition
[7] alternate candidate material is 304L stainless steel

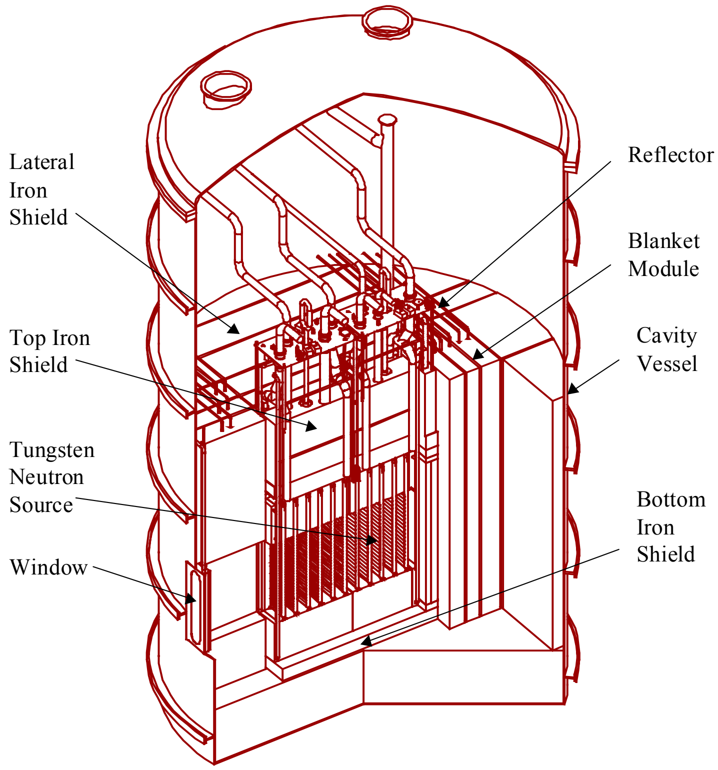

Figure 1 – APT Target/Blanket Assembly

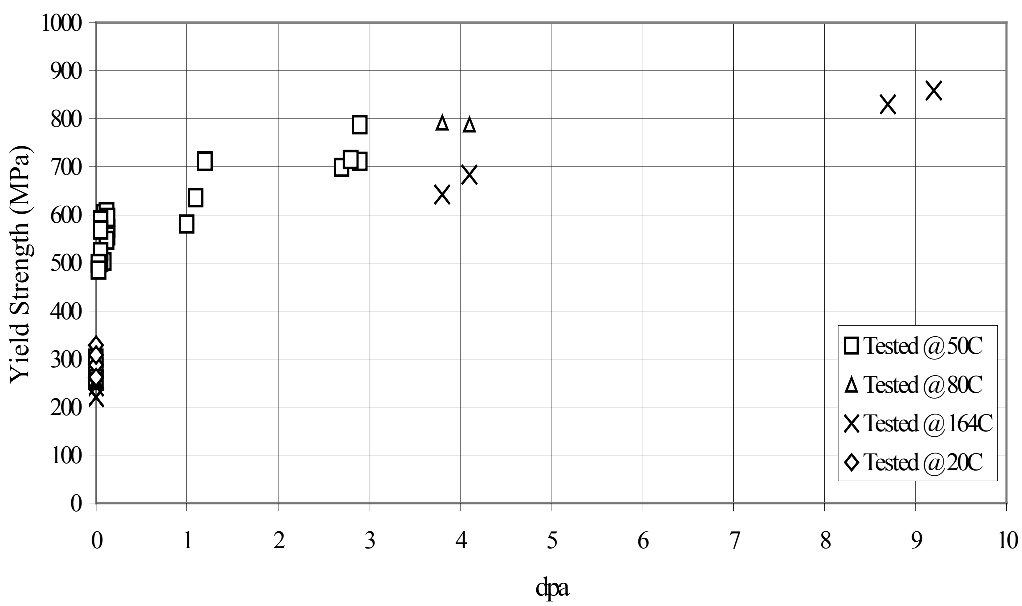

Figure 2 – Yield strength of 316L stainless steel after proton/neutron irradiation in the LANSCE facility at temperatures up to 164C.

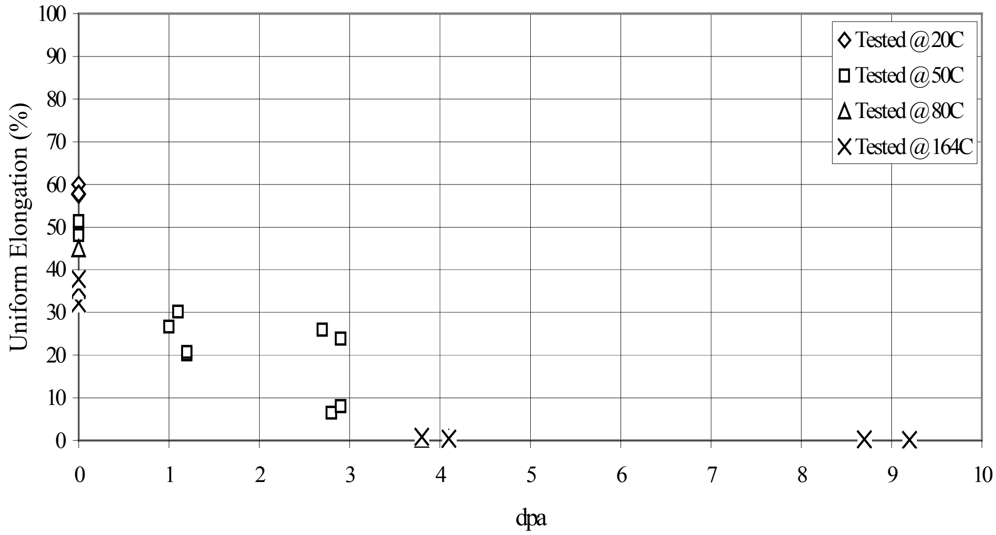

Figure 3 – Uniform elongation of 316L stainless steel after proton/neutron irradiation in the LANSCE facility at temperatures up to 164C.

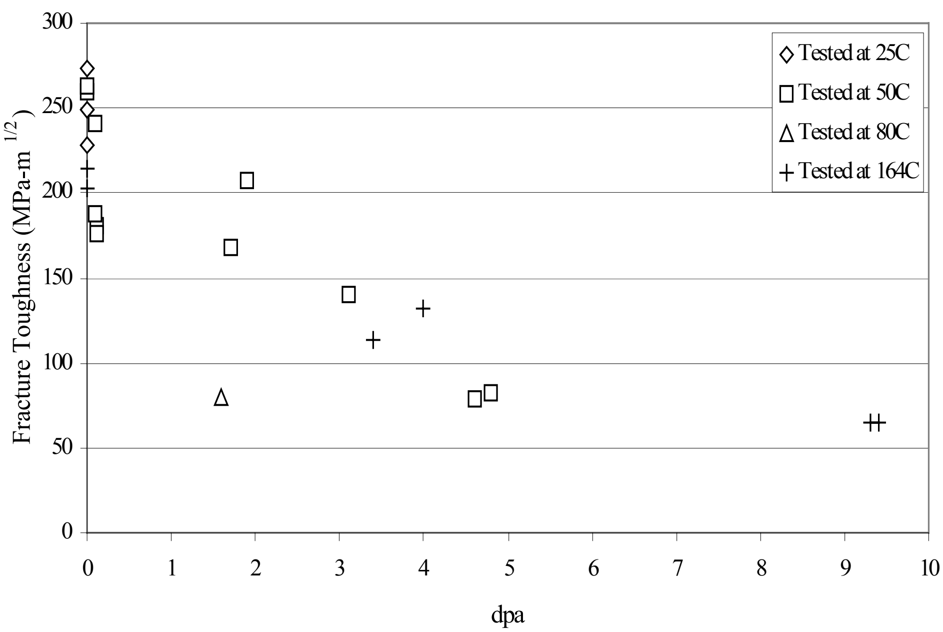

Figure 4 – Fracture toughness of 316L stainless steel after proton/neutron irradiation in the LANSCE facility at temperatures up to 164C.

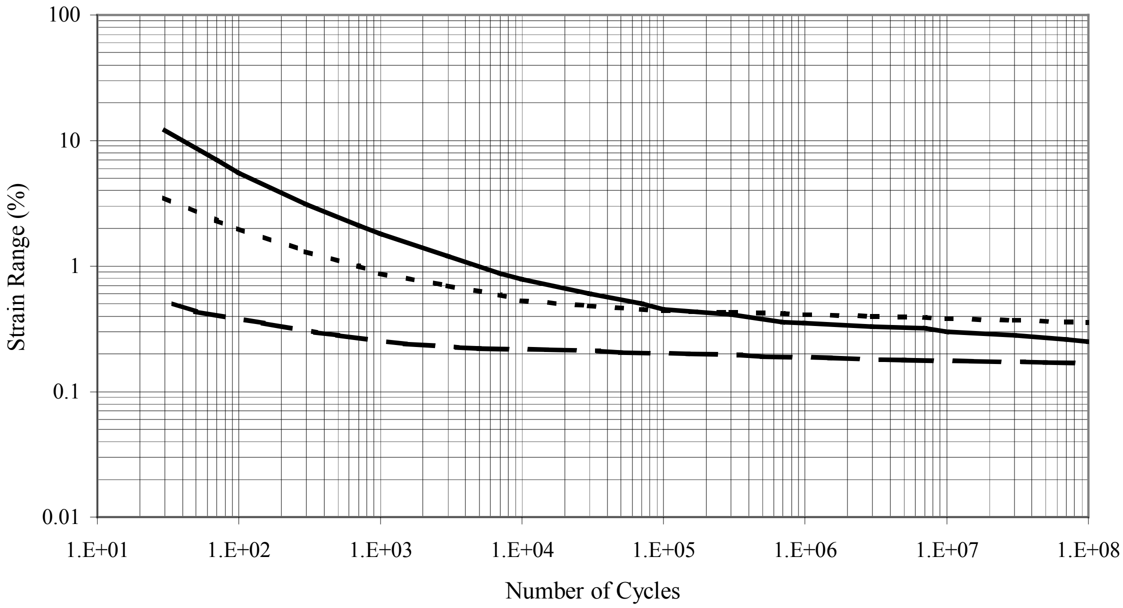

Figure 5 – Generalized fatigue behavior of an irradiation-hardenable alloy (e.g., 316L stainless steel) and design curve based on APT Supplemental Structural Design Requirements.

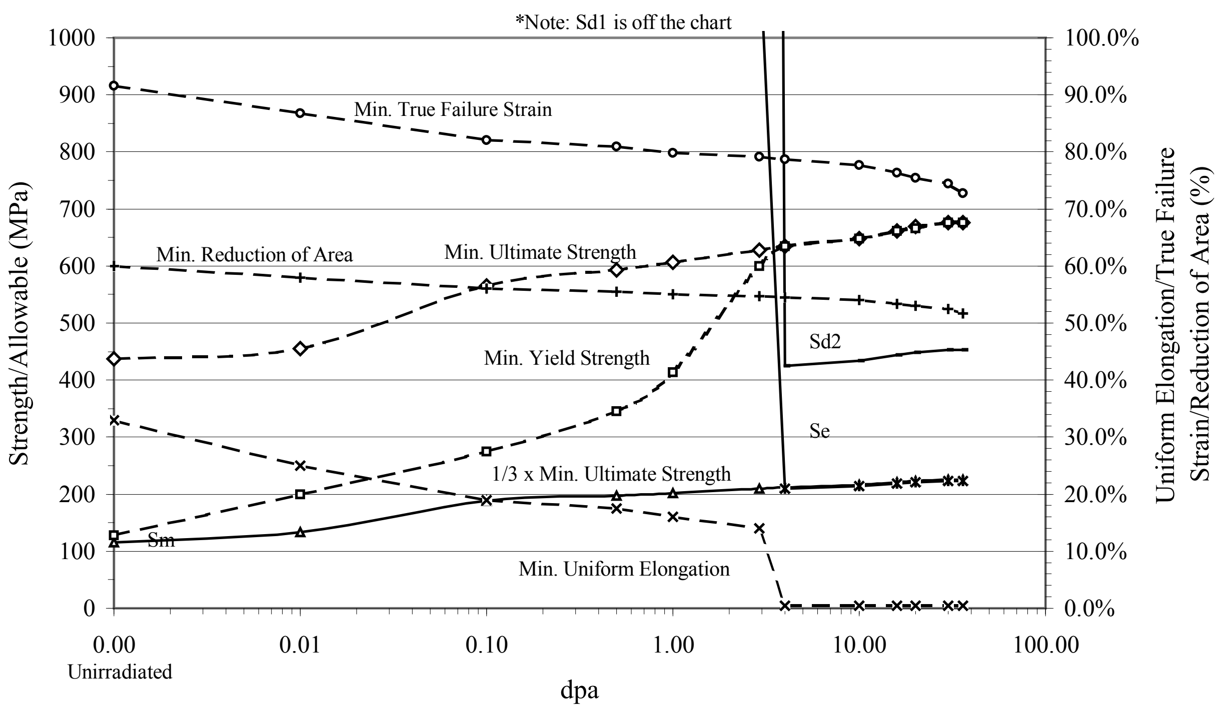

Figure 6 – Basic minimum tensile properties and design stress allowables for proton/neutron-irradiated annealed 316L stainless steel.

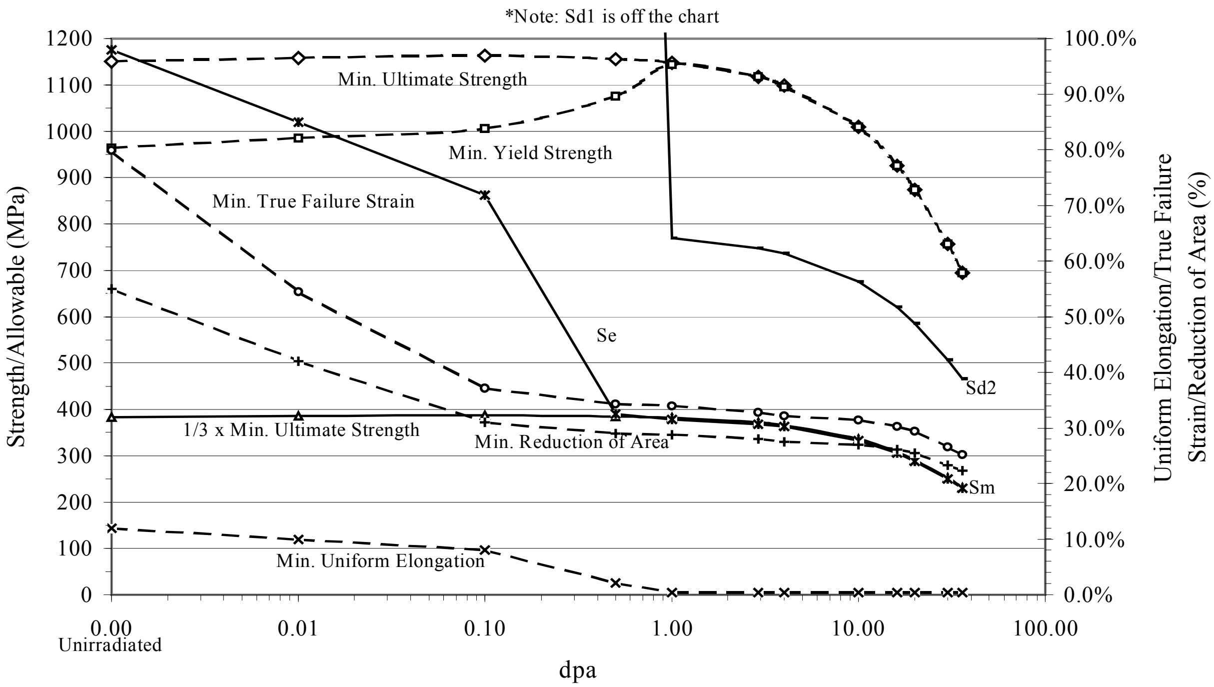

Figure 7 – Basic minimum tensile properties and design stress allowables for proton/neutron-irradiated precipitation-hardened Alloy 718.